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CAREM PROJECT:INNOVATIVE SMALL PWR

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ABSTRACT

CAREM, an Argentinean project, consists on the development, design and construction of a small nuclear power plant. CAREM is an advanced reactor conceived with new generation design solutions and standing on the large experience accumulated in the safe operation of Light Water Reactors. This paper summarizes the design of the prototype CAREM and associated development activities.

1.INTRODUCTION

CAREM is a CNEA (Comisión Nacional de Energía Atómica) project, which is jointly developed by INVAP. CAREM is a project for an advanced, simple and small nuclear power plant, conceived with new generation design solutions and standing on the large world wide experience accumulated in the safe operation of Light Water Reactors. The CAREM is an indirect cycle reactor (100 MWt, approx. 27 MWe) with some distinctive features that greatly simplify the reactor and also contribute to a higher level of safety:

- Integrated primary cooling system.
- Primary cooled by natural circulation.
- Self-pressurized.
- Safety systems relying on passive features.

The CAREM NPP is an integrated reactor. The whole high-energy primary system -core, steam generators, primary coolant and steam dome- is contained inside a single pressure vessel. The flow rate in the reactor primary systems is achieved by natural circulation. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form pressure-losses, producing a flow rate in the core that allows to have a sufficient thermal margin to critical phenomena. The driving force for the coolant's natural convection is produced by the location of the steam generators above the core. The coolant acts also as moderator. Self-pressurization of the primary system in the steam dome is the result of the liquid-vapour equilibrium. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Due to self-pressurization, the core outlet bulk temperature corresponds to saturation temperature at primary pressure. Heaters and sprinkles typical of conventional PWR's are thus eliminated. The main criteria used in the design of safety systems were simplicity, reliability, redundancy and passivity. Special emphasis has been put on minimizing the dependence on active components and operators' actions.

This document contains a brief technical description of this project.

2. TECHNICAL DESCRIPTION

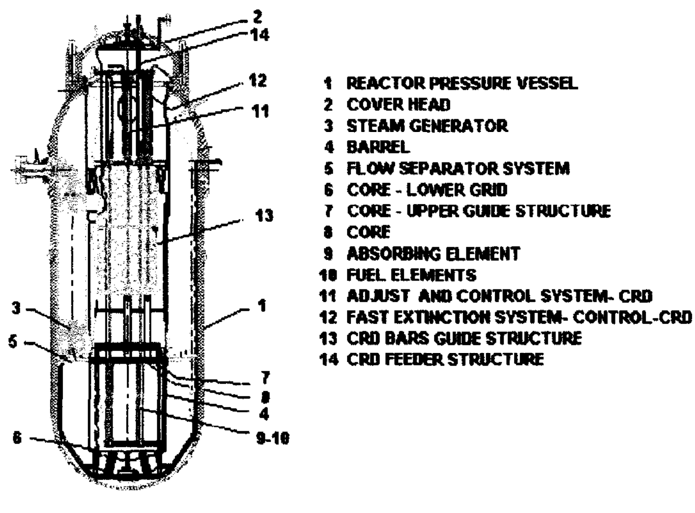

Fig.-1 Reactor Pressure Vessel

2.1. Primary system

The core has 61 fuel assemblies (FA) of hexagonal cross section. The fuel is UO_2 enriched at 1.8 and 3.4%. An 8% weight of Gd_2O_3 is used as burnable poison to flatten the power distribution along the fuel cycle. Fuel cycle can be tailored to customer requirements, with a reference design of 330 full-power days and 50% of core replacement.

Each absorbing element (AE) consists of a cluster of rods linked by a structural element (namely “spider”), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes, at 18 positions in the FA not occupied by fuel rods. The absorbent material is the commonly used Ag-In-Cd alloy. Absorbing elements (AE) are used for reactivity control during normal operation (Adjust and Control System), and to produce a sudden interruption of the nuclear chain reaction when required (Fast Extinction System). Chemical shim is not used for reactivity control during normal operation.

Twelve identical ‘Mini-helical’ vertical steam generators, of the “once-through” type are placed equally distant from each other along the inner surface of the Reactor Pressure Vessel (RPV) area. They are used to transfer heat from the primary to the secondary circuit, producing dry steam at 47 bar, with 30°C of superheating.

The location of the steam generators above the core produces natural circulation in the primary circuit. The secondary system circulates upwards within the tubes, while the primary does so in counter-current flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system guarantees that the entire stream of the primary system flows through the steam generators. In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized by changing the number of tubes per coil layer. Thus, the outer coil layers will hold a larger number of tubes than the inner ones. For safety reasons, steam generators are designed to withstand the pressure from the primary up to the steam outlet / water inlet valves even without pressure in the secondary.

The natural circulation of the coolant produces different values of flow rate in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained.

Due to the self-pressurizing of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions, it has been proved sufficient margin to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit combined with the self-pressurization features make this behaviour possible with minimum control rod motion.

It concludes that the reactor has an excellent behaviour under operational transients.

2.2 Safety systems

The First Shutdown System (FSS) is designed to shut down the core, when abnormal or deviated from normal situations occur, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping a total of 25 neutron-absorbing elements into the core by the action of gravity. Each neutron absorbing element is a cluster composed of a maximum of 18 individual rods which are put together in a single unit. Each unit fits well into the fuel assembly guide tubes.

Hydraulically moved AE (CRD) avoid the use of mechanical shafts passing through, or the extension of the primary pressure boundary, and thus eliminates possibilities of big Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV. Their design is an important development in the CAREM concept. Six out of twenty-five CRD, which simplified operating diagram are shown in figure-2, are part of the Fast Extinction System, and are kept in the upper position during all normal operation with an external hydraulic circuit that pumps water to a lower chamber of a piston/cylinder assembly. Once at the top, the piston partially closes the outlet orifice and reduces the water flow to a leakage.

Both device types, perform the SCRAM function by the same principle: “rod drops by gravity when flow is interrupted”, so malfunction of any powered part of the hydraulic circuit (i.e. valve or pump failures) will cause the immediate extinction of the reactor. CRD of the Fast Extinction System are designed using a large gap between piston and cylinder in order to obtain a minimum dropping time thus taking few seconds to insert absorbing elements completely inside the core. The CRD of the Adjust and Control System is designed to guarantee that each pulse will produce only one step, so manufacturing and assemblies' allowances are stricter and clearances are narrower'. In this case there is not stringent requirement on dropping time.

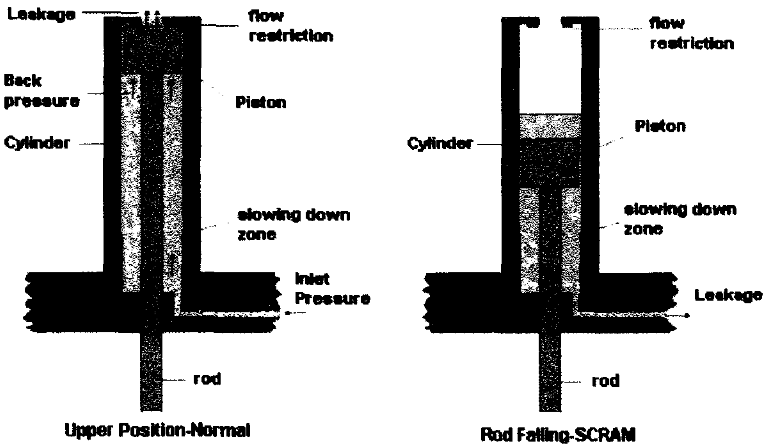

Fig.-2 Simplified Operating Diagram of Hydraulic Control Rod Drive Fast Extinction System

A gravity-driven injection system of borated water at high pressure makes up the second shutdown system. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. The system consists of two tanks of 2 m^3 located in the upper part of the containment. Each of them is connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank produces the complete shutdown of the reactor.

The Residual Heat Removal System has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected via piping to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside of the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the pressure suppression pool of the containment.

The Emergency Injection System prevents core exposure in case of LOCA. In the event of such accident, the primary system is depressurized with the help of the emergency condensers to less than 15 bar, with the water level over the top of the core. At 15 bar a low pressure water injection system comes into operation. The system consists of two tanks with borated water connected to the RPV. The tanks are pressurized to 21 bar, thus when during a LOCA the pressure in the reactor vessel reaches 15 bar, the rupture disks break and the flooding of the RPV starts.

The primary system, the reactor coolant pressure boundary, and important auxiliary systems are enclosed in the primary containment, a cylindrical concrete structure with an embedded steel liner. The primary containment is of pressure-suppression type with two major compartments: a drywell and wetwell. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition floor and cylindrical wall separate the drywell from the wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber.

Three safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the core power and the power removed by the systems. Each valve is capable of producing 100% of the necessary relief. The blow-down pipes from the safety valves are routed to the pressure suppression pool.

The responses of the plant to blackout, LOCA and main steam pipe break is described. The blackout is one of the events with major contribution to core meltdown probability in a conventional light water reactor. On the contrary, in the CAREM NPP, the feedback coefficients will produce the self-shutdown of the nuclear reaction. The cooling of the core and the decay heat removal are guaranteed without electricity by the passivity of safety systems. At the same time, the power cut off produces the interruption of the feed water to the hydraulically driven shutdown system, and thus produces the insertion of the absorbing elements into the core. The decay heat will be removed by the Residual Heat Removal System.

Since only small LOCA's are possible, and due to the large water inventory of the primary system, there is a long time span between the initiation of the LOCA and core exposure, in comparison with conventional PWR's. The largest break in the primary system allows some minutes of depressurization before the Emergency Injection System comes into operation with the RPV pressure at 15 bar and the core fully covered. It produces a transient that can be easily handled by the safety systems due to the small water inventory of the steam generators in the secondary side and the large water inventory of the primary system.

2.3 Plant design

The CAREM nuclear island is placed inside a containment system, which includes a pressure suppression feature to contain the energy and prevent a significant fission product release in the event of a postulated design basis accident.

The building surrounding the containment has been designed in several levels and it is placed in a single reinforced concrete foundation mat. It supports all the structures with the same seismic classification, allowing the integration of the RPV, the safety and reactor auxiliary systems, the spent fuels pool and other related systems in one block. The plant building is divided in three main areas: control module, nuclear module and turbine module.

Finally, CAREM NPP has a standard steam cycle of simple design

2.4 Advantages of CAREM design

Technical and economical advantages are obtained with the CAREM design compared to the traditional design:

- No large LOCA has to be handled by the safety systems due to the absence of large diameter piping associated to the primary system. The size of maximum possible break in the primary is 38 mm.
- Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or severe accidents.
- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The large water volume between the core and the wall leads to a very low fast neutron dose over the RPV wall.
- Eliminating primary pumps and pressurizer results in lower costs, added safety, and advantages for maintenance and availability.

CAREM-25 may be seen, as a product, under several points of view. First, as a National Project part of Argentine Nuclear Policy, assures the availability of updated technology in the mid-term. This implies working with the technology already acquired in design, construction and operation of Research Reactors and operation and improvements of Pressurized Heavy Water Reactors, and developing advanced design solutions. Second, as a "Bridge Project", connecting R&D nuclear activities with full scale Nuclear Power Plants. It allows the development of human resources, technical/industrial capabilities and licensing and management infrastructure. Finally, as a very Small Nuclear Power Plant, well suited to provide electricity in isolated areas, competes in restricted isolated markets. In addition, CAREM may be implemented for co-generation applications as seawater desalination or process heat production.

3. DEVELOPMENT ACTIVITIES

The CAREM is an innovate reactor, so many developed activities have been involved. The effort has been focused mainly on the nuclear island (inside containment and safety systems). This comprises mainly: the Reactor Core Cooling System (RCCS), the Core and Fuel Assembly, the Reactor Pressure Vessel Internals (RPVI), and the First Shutdown System (FSS).

Related to the RCCS modelling and qualification are boosted by the testing performed in a High Pressure Natural Circulation Rig (CAPCN), covering Thermal Hydraulics (TH), reactor control and operating techniques. This facility may also be used to test the Second Shutdown System, some in-vessel instrumentation probes and special feed-throughs from in-vessel wired signals. The CAPCN rig reproduces all the dynamics phenomena of the RCCS, except for three-dimensional effects.

Most of the tests consist of an initial self-steady state in which a pulse-wise perturbation induces a transient. In this case the perturbation is a thermal unbalance as severe as possible for operational transients: thermal power is increased 12 KW (about 5%) during 150 seconds. As it may be seen, primary pressure and circulating flow evolve mildly, with increases below 2 and 3% respectively, and primary temperatures hardly notice the perturbation. Therefore steam generation remains quite stable during the whole transient, a remarkable feature for a Steam Supply System

The Core Design involves different aspects i.e. study of thermal limits, neutron modelling, fuel assembly design and structural mechanical design. A set of experiments is being conducted in a high-pressure rig at nominal and out of nominal conditions. Neutronic modelling needs are covered by benchmark data available world-wide. As for Fuel assemblies design, CNEA has vast experience in the technology of nuclear fuels. A series of hydrodynamic and structural tests are carried out at two (low and high) pressure rigs.

Mass flow rate in the core of the CAREM reactor is rather low compared to typical light water reactors and therefore correlations or experimental data available for thermal limits are not completely reliable in the range of interest. Thus analytical data needed to be verified by ad-hoc experiments. Experimental program to generate a substantial database to develop a prediction methodology for Critical Heat Flux were performed at Institute of Physics and Power Engineering (IPPE) located at Obninsk (Russian Federation).

Related to mechanical design (structural, dynamic, seismic, etc.) of the core and other RPVI, different mock-up facilities are being constructed. They represent sections of the core, and include one vertical full-scale model with supporting barrel and its kinematics chain.

An important experimental plan is underway for the FSS, or more specifically the CRD. The first series of tests are conducted in the Cold Low Pressure Rig (CEM) and second series in the High Pressure Rig for CRD Test (CAPEM).

4. CONCLUSIONS

CAREM is a CNEA project, which is jointly developed by INVAP. The project consists on the development, design and construction of a prototype small nuclear power plant. The CAREM is an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the reactor and also contribute to a higher level of safety: integrated primary cooling system, self-pressurized, primary cooling by natural circulation, safety systems relying on passive features. Therefore, many technical and economical advantages are obtained with the CAREM design compared to the conventional designs.

CAREM, is a very small nuclear power plant, well suited to provide electricity in isolated areas, in addition it may be implemented for co-generation applications as seawater desalination or process heat production.